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Abstract

GRAIN SIZE AND MICROSTRUCTURE EFFECTS ON IRRADIATION HARDENING OF LOW CARBON STEEL FOR REACTOR TANKS

by

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The grain size effect on irradiation hardening of different types of steel has been studied extensively during the past few years. However, the evidence obtained was not sufficient to answer completely to the question.

The experiments described here were performed to give a new evidence on the influence of the grain size parameter on irradiation hardening of a given type of low carbon steel for the reactor tank at integral fast flux > 10^{19} nvt and irradiation temperature below 100°C. As a part of the VISA 2/1 Project these investigations were carried out in collaboration with the Saclay Nuclear Research Center.

At the same time the experimental procedures used to obtain different grain size specimens gave different microstructures of the steel. Thus, it was necessary to deal not only with different grain sizes, but also to take into account possible influence of the microstructure obtained. Actually, it was found that the behaviour of the specimen during irradiation was more affected by microstructure than by grain size.

The most important results obtained could be summarized as to indicate: (1) different grain size specimens, having ferrite-perlite microstructure gave the same increase in a yield stress, (2) specimens with cementite precipitated at the grain boundaries have shown a much more pronounced embrittlement; after irradiation all of them fractured in a brittle manner before the plastic region has been reached, (3) the specimens treated in austenitic range to obtain large grain sizes embrittled less than the specimens deformed at room temperature and treated in ferritic range, the elongation to fracture being reduced much more in the latter case. This is also followed by a more pronounced change in the shape of the stress-strain curve of the specimens treated in ferritic range.

The annealing experiment was performed at 150°C. In general, ferrite-perlite microstructure specimens have shown a further small increase of the yield and ultimate tensile stress.

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Abstract

COMPLEX EXPLOITATION OF URANIUM IN A DEPOSIT INCLUDING THE LOW-GRADE ORES

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To solve the problem of exploitation of uranium of a given deposit, the detailed investigation was required to study the economically feasible extraction of metal from the low grade ores.

In the exploitation of uranium deposits, high grade ores are processed by the "agitation" in hydrometallurgical plants. Low grade ores, unavoidably accompanying the high grade ones, are partly left in situ and partly stocked on the surface. Due to the low concentration of uranium, these low grade ores - waste - have not been treated for the extraction of uranium up to the present time.

The intention of our studies has been to determine higher economy of the process of uranium extraction from the low grade uranium ores. The crushed low grade ores were treated by circulation of the corresponding solutions, to prevent large scale investment costs, as well as the costs necessary for more classical ore processing in the plants.

The results of the laboratory scale investigations have been demonstrated, as well as the large scale field investigations.

Four different ore samples have been selected in the laboratory studies. The results of these investigations and the factors which could have influence on the uranium leaching by this process have been discussed. The results of these studies have demonstrated lower or even the same uranium leaching comparing with the process of "agitation".

Large-scale investigations have been carried out in the mine. Special attention has been paid to the extraction of uranium at the lowest possible costs. The tests have been carried out in the pit - in situ and outside during the leaching process - "in heap", with larger quantities of ores. These experiments have confirmed the laboratory results and pointed out the feasibility of the application of leaching in situ and "in heap".

The results of described investigations point out the possibility of the extraction of uranium from the low grade ores at acceptable costs.

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Abstract

INDUSTRIAL APPLICATION OF CATALYTIC PRECIPITATION OF URANIUM

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The recent investigations have verified that the process of reduction of U (VI) to U (IV) is the catalytic one in which the selection and form of catalyst have a large influence on the feasibility of the process.

It has been demonstrated that the process of reduction of U (VI) to U (IV) from the water solutions of sodium carbonate is best carried out with UO_2 as a catalyst, which is at the same time produced by the reaction itself.

Laboratory and pilot plant experiments with the nickel wire pointed out that the uranium oxide is deposited on the surface of the wire. The thickness of the deposited oxide ranged up to and more than 5 millimeters, and the density ranged from 4 to 5 g/cm^3. The process of reduction was developing with catalyst deposited in this way. To check catalytic abilities of uranium dioxide without the presence of nickel wire the reduction process has been studied and the results of the application of uranium-dioxide in the form of powder and "rachige" rings have been demonstrated.

The activation of catalyst at the increased temperature, in hydrogen stream have been illustrated. This kind of activation enables the application of such catalyst for the precipitation of uranium from the alkaline solutions at a temperature below 100°C, at normal hydrogen pressure.

Based on the results of the laboratory pilot plant experiments, the construction of full scale plant for continuous uranium precipitation from carbonate leaching solutions has been undertaken.

The description of the plant and the basis for the selection of processing, together with results and analysis was also presented.

The effect of precipitation of uranium as a function of the application of uranium dioxide as a catalyst retention time, concentration of agents and uranium, method of agitation, temperature and pressure has been analysed.

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Abstract

URANIUM DEPOSITS IN TERTIARY VOLCANITES OF NORTH-EASTERN MACEDONIA

by

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In the volcanic area of Kratovo and Zletovo /NE Macedonia/, already known for the rich lead-zinc deposits, the important uranium deposits have been discovered recently. This area is located in the border zone of the Dinaric orogene belt and the internides, i.e., in the axial part of the Alpine orogene of this area of Balcan Peninsula.

The general geology is relatively simple and the main formations are paleozoic and lower paleozoic crystaline schists, tertiary volcanites and tertiary sediments.

The volcanites cover an area of more than 1500 km^2 and are connected with a large, deep fault zone, located near the bordering area. This zone is in the north also marked by the similar volcanic complexes.

The structure has been formed under the influence of a disjunctive system, created during a number of phases of Alpine orogenesis. The fault zone extends basicaly to NNW-SSE /N-S/ and ENE-WSW /E-W/, with very steep, almost vertical dips. The first system is a dominant one and represents the basic structural element of this part of Alpine orogene. The basic network of the intersections of these dislocations is between few hundred meters up to two kilometers, and the width of the tectonic zones along the main structures amounts

up to and more than 100 meters.

Multiphase volcanic eruptions produced several rock types: amphibole-biotite andesites and their agglomerated tuffs, the ignimbrites of dacitic and quartz-latitic origin, dacitic stratified tuffs, amphibole-biotite andesites and quartz-latites and labradorite augite andesites and associated tuffs. The greatest area is covered by the ignimbrites of dacitic and quartz-latitic origin.

The mineral and ore occurences are connected to the fault and tectonic zones. The uranium mineralization could be divided into two types, as an independent mineralization and in the connection with Pb-Zn mineralization.

The characteristic representative of the first type is the Zletovska Reka deposit, discovered by the middle of 1962. The deposit is located in volcanites in the vicinity of their contact with the crystaline schists of the internides. The local structures are represented by the intensively, tectonicaly and hydrothermaly, modified zones, extending E-W, 20-100 meters wide and more than 3 kilometers long. These zones represented the ore controling features.

One of these zones has been explorated about 300 meters along the strike and up to 150 meters along the dip. The uranium ore bodies are only slightly oxidized and in the lower levels are composed exclusively of the pitchblende.

The reserves in the explorated part of the zone amount about 300 000 tons of ore, with a mean uranium content of 0.15 % of U_3O_8.

The discovery of uranium in such structures and formations is of basic importance, due to the large potential possibilities of discovery of similar deposits in the younger volcanites of the Alpine orogene.

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Abstract

A Crystal Diffraction Spectrometer for Non-destructive Burn-up Analysis

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A bent-crystal gamma-ray spectrometer has been built for burn-up measurements on reactor fuel elements. It will mainly be used adjusted to the Du Mond geometry. According to focusing properties a line-shaped source must then be used and consequently a very narrow part of the fuel element will be analyzed. The extension of the source is defined by slits in the radiation shield in front of the fuel element. The combined effect of slit geometry and gamma-ray self-absorption in the fuel element results in a suitable approximated line source.

The spectrometer can easily be changed for use also in the reversed (Cauchois-) geometry. In this case a larger part of the fuel element is studied, the size of which being mainly determined by the effective area of the diffraction crystal. In both cases a NaI-crystal is used as detector.

The bending radius of the diffraction crystal is four meters. Diffraction takes place against the (310)-planes in a 80 x 80 x 3 mm quartz crystal. The fuel element is placed in a radiation shield consisting of a transportable gas-sealed container, where it can be rotated and elevated in front of the source-defining slits. Consequently, a detailed analysis over the whole element surface is possible.

Determination of the burn-up is based on intensity measurements of diffraction peaks from fission product gamma lines. From measurements performed at our one-meter bent-crystal spectrometer the required gamma intensities of some selected fission products have been estimated. Calculations show that the intensities from the chosen gamma lines will be sufficient for obtaining diffraction peaks of reasonable statistics.

Work is going on aiming at the use of bent germanium crystals. This would increase the efficiency of the spectrometer by about one order of magnitude.

This work is performed under contract nr 163/RB with the International Atomic Energy Agency, Vienna.

1) Chalmers University of Technology, Göteborg

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Abstract

Fuel Cycle Physics and Economics for Heavy Water Reactors

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Sweden is engaged in the development of heavy water moderated reactors of the pressure vessel type using slightly enriched uranium for normal operation. The present reactor programme, based on the prospective energy needs over the next two decades and the most economical way of developing the overall power system, is briefly outlined.

From the aspect of fuel cycles two important items influence the choice of a particular reactor type, namely long range fuel utilization and fuel cycle economics. The prospective consumption of natural uranium in Sweden is compared for different reactor types, including the heavy water moderated Marviken reactor type, the light water moderated advanced converter reactor and the fast breeder reactor. Fuel cycle costs are compared with special regard to the effects of feed cost, fabrication and reprocessing costs, uranium and plutonium credit, and burnup.

The characteristics of current Swedish fuel cycle policy are discussed. This involves consideration of the use of natural or enriched fuel, the need for chemical reprocessing of spent fuel, and the possibility of plutonium recycling. Presently adopted fuel cycles for the 200 Mw elect. Marviken reactor, which has entered the construction phase, and for the 350 Mw elect. PHWR conceptual design will be discussed. The Marviken reactor is being designed for on-power refuelling, whereas PHWR uses refuelling at shut-down. Both reactors use fuel management as a means of reactivity

1) AB Atomenergi, Stockholm

control and power shaping. The implications of these features for reactor operation and fuel cycle costs are evaluated. The operational problem of approach to fuel cycle equilibrium is indicated.

It is shown that there is at the present time an economic incentive to use slightly enriched fuel in the heavy water reactors under consideration, whereas no chemical reprocessing of spent fuel is required. As a possible development the use of plutonium recycle is considered. The value of plutonium and the possibility of self-sustained recycling with natural uranium feed are discussed. Different refuelling schemes, plutonium feed compositions and cost assumptions are included. Different heavy water reactor types are compared, including cooling with pressurized water, saturated steam and internally superheated steam. Optimization of reactor parameters for the use of plutonium is considered.

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Abstract

A Reprocessing Alternative for Zircaloy Clad Natural or Slightly Enriched Uranium Oxide Fuel Elements

E. Haeffner [1], Å. Hultgren [1], A. Larsson [1]

The main interest for processing spent fuel elements containing natural or slightly enriched uranium is the recovery of the plutonium produced during irradiation. The recycling of depleted uranium in thermal power reactors seems at present to be of low economical value. In the reprocessing cycle it is therefore not necessary to decontaminate the uranium further than required for its storage without increasing the cost for radiation shielding.

The reprocessing method described here is intended for the treatment of uranium oxide fuel canned in zirconium alloys and irradiated to a burn-up of the order of 10,000 Mwd/t. The decanning is carried out by high temperature oxidation of the zirconium followed by a dissolution of the oxide core in nitric acid. Plutonium is separated from the uranium and fission products by an amine extraction process. The uranium is decontaminated to an adequate degree with tributyl phosphate extraction.

An economical evaluation of a plant of 1 t/day capacity is made.

1. The chemical head-end method usually proposed for Zircaloy clad fuel (the Zirflex process) produces considerable amounts of liquid intermediate radioactive wastes, which cause high storage costs. The proposed Thermox process has the advantage that only small amounts of solid waste emerges from the decanning step.

1) AB Atomenergi, Studsvik and Stockholm

Whole fuel element assemblies are introduced into a vertical furnace tube where the zirconium is oxidized in air and steam at 800 °C. Subsequently the uranium oxide is dissolved in dilute nitric acid. The zirconium oxide appears as solid waste in a volume one or two orders of magnitude less than the corresponding liquid Zirflex waste.

2. Tertiary amines used as solvents for extraction are characterized by a high selectivity for plutonium in its 4-valent state. This property is unchanged also after a high radiation exposure.

In this study two alternative chemical flowsheets are presented. After a primary plutonium extraction with a solvent containing 10% trilaurylamine (TLA) in tertiary butylbenzene and subsequent scrubbing, the organic plutonium stream in the first alternative passes through a silica gel column for plutonium reextraction and concentration. The plutonium eluted from this column is further purified in an anion exchange cycle.

The proposed silica gel process was tested in the laboratory with actually expected plutonium concentrations.

In the second alternative, the plutonium, after a primary extraction and scrubbing, is reextracted with acetic acid. A second plutonium extraction with 20 % TLA is followed by a direct precipitation of plutonium from the organic solution with oxalic acid.

The uranium is decontaminated in a separate tributyl phosphate cycle to a degree sufficient for subsequent handling and storing.

3. The plant layout and cost analysis are based upon the second chemical flow-sheet alternative.

Special equipment is required for the head-end process while equipment of more conventional design is assumed for the rest of the plant.

The cost estimate includes not only the main plant building but also general services and utilities required for operation of the plant.

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Abstract

Some Critical Problems in Recent Swedish Work on Cladding Materials

G. Östberg [1], H.P. Myers [1]

Swedish research work on cladding materials has in recent years essentially been concerned with zirconium alloys and stainless steel for water cooled fuel elements. The major part on zirconium has been done for the Ågesta reactor where the operating temperature is about 250° C (pressurized water). The aim of the studies of stainless steel is to find a canning material for the Marviken superheater elements which are to produce 500 °C steam.

Concerning zirconium alloys it has been found that Zircaloy-2 in the unirradiated condition might behave in a brittle manner if strained slowly. This embrittlement, which is associated with other irregularities in deformation behaviour due to metallurgical changes, is probably caused by deformation-induced hydride precipitation. Studies of texture have resulted in a suggestion that this property should not only be carefully observed but also intentionally employed for obtaining optimum strength and ductility of tubular components. After irradiation of Zircaloy-2 with varying hydrogen contents a yield point has been found. Comparative studies of dislocation structures in zirconium after different treatments have been made.

The effort done on corrosion of Zircaloy-2 has been limited to some exploratory studies of the electrochemistry of water corrosion and a series of hydrogen absorption measurements, in and out-of-pile, on Ågesta canning material of various sources and treatments. Parallel to this work studies on the metallography of Zircaloy have been made, providing information on the composition of the second phase in Zircaloy-2 and the microstructural appearance of the hydride as well as its solubility.

1) AB Atomenergi, Stockholm and Studsvik

For superheat element canning a number of alloys of stainless steel type, mainly austenitic alloys with increased nickel content, are being extensively studied. Besides the properties generally considered, i.e. corrosion resistance and mechanical strength and ductility, a few critical properties are examined with particular care. These are, on the corrosion side, pitting and chloride-induced stress corrosion, both transgranular and intergranular, the latter in high temperature steam. On the mechanical side particular attention is given to strain fatigue and other processes leading to reduced ductility.

The microstructural background to these phenomena is followed and a few observations on segregation of minute amounts of boron, believed to be related to embrittlement by thermal neutrons, have been made.

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Abstract

Irradiation Behaviour of Swedish Steels for Reactor Pressure Vessels

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Present Swedish power reactors are designed with steel pressure vessels and the embrittlement of pressure vessel steels by neutron irradiation has therefore been a subject for detailed study. Results from some thirty irradiation experiments in the R2, DR2 and DR3 reactors are now available. A systematic study of the effects of irradiation dose (10^{18}-10^{20} nvt>1 Mev) and irradiation temperature (50-300°C) has been made on three C-Mn mild steels, including ASTM A212B; three low alloy Mn-Mo steels including ASTMA302B; a B-Mo alloy steel and two versions of a martensitic austenitic 13 Cr 6 Ni steel. These steels have been irradiated in the conventional conditions as produced by the manufacturer but, in co-operation with industry, metallurgical variables including the effects of grain size, deformation ageing and quenching and tempering have also been considered in certain cases.

The irradiation changes in the hardness, impact energy to fracture and tensile strength of these steels have been measured and show different steels, or different conditions of the same steel, to exhibit widely differing susceptibilities to irradiation embrittlement. For this work conventional testing techniques have been used but in addition a miniature Charpy V specimen has been developed. This has been found to be eminently satisfactory for the irradiation of two or more steels under identical conditions of temperature and neutron flux in rigs of limited volume. This work allows both an assessment of the irradiation behaviour of Swedish pressure vessel steels at present available as well as a comparison with certain standard American qualities.

The brittle ductile transition temperature of a pressure vessel steel is affected by factors and treatments other than neutron irradiation and an assessment of a steel's suitability for use in a reactor vessel must include careful consideration of aspects such as stress relieving, deformation ageing, heat affected zones around welds and the properties of the weld metal. The significance of irradiation effects as well as other metallurgical variables influencing the brittle ductile transition temperature and affecting the use of steel in reactor pressure vessels will be discussed.

1) AB Atomenergi, Studsvik

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Abstract

Sintered Aluminium Products for Organic Reactor Application

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The Danish Atomic Energy Commission has undertaken a study of an organic-cooled, heavy-water-moderated reactor. Part of this project has been the investigation of sintered aluminium products (SAP), which have been selected for pressure tubes and canning material for the uranium-dioxide fuel.

Sintered aluminium products containing 7 to 11 wt% of aluminium oxide have a satisfactory creep strength at the reactor temperature, but the elongation at creep rupture is very small. Tensile testing of specimens subjected to creep stresses for extended periods has, however, shown satisfactory elongation.

Vacuum degassing of SAP stabilizes the material; thus fissure formation in the microstructure at elevated temperatures is avoided. Extended heat treatment of SAP at 500°C followed by microscopic examination has shown that vacuum-degassed material is completely stable.

Neutron irradiation at 40° and 270°C has a slight effect on the mechanical properties of SAP as examined by tensile and impact testing.

The compatibility between SAP and uranium dioxide is satisfactory. The compatibility between SAP and uranium monocarbide is unacceptable, unless a barrier layer is introduced. Flash welding of SAP canning tubes results in end closures of acceptable tightness and strength. A welding method utilizing a rotating arc is promising for the closing of SAP tubes with aluminium end caps.

Finned tubes of SAP according to the fuel-element design have been successfully extruded by a British manufacturer, and non-destructive methods for measuring the internal diameter and wall thickness of the tubes have been developed.

Alloy development work has resulted in a new type of aluminium/aluminium-oxide material manufactured by mechanical mixing of the constituents. The new alloys have properties comparable to those of SAP, although they are made by a completely different process.

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Abstract

TRANSPORT OF IRRADIATED NUCLEAR FUEL ON THE EUROPEAN CONTINENT

by

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This paper gives an overall description of the present status of irradiated fuel transport in Continental Europe, its administrative, technical and economic aspects and its influence on fuel cycle costs. Transport of irradiated fuel, of processed fissile material and possibly of waste material between different countries on the Continent will become routine within a few years.

Since 1963, MTR fuel elements have been transported routinely from the Continent for processing. Bulk shipments from the first power reactors have not yet begun but it is anticipated that 270 tonnes of irradiated fuel will be unloaded from Continental power reactors before 1968. A major part of this fuel is of sufficiently high value to economically justify its reprocessing and thus increasing fuel shipments are anticipated.

* One of the joint undertakings of the O.E.C.D. European Nuclear Energy Agency.

A brief survey is made of national legislation and transport regulations, as well as procedures necessary for transport approval as laid down in a number of European countries. International regulations, and their underlying safety principles, are mentioned and their importance in achieving uniform application and uniform cask design is discussed.

Casks conforming with the regulations have been manufactured for a number of Continental reactors. Parameters governing cask design arise from regulatory requirements such as fuel temperature limitation during accident conditions which tend to increase overall transport costs. On the other hand, improvements in cask design or in cask utilisation, for example where casks are developed to serve more than one reactor, can achieve a reduction in transport costs. Factors affecting cask payload are described in view of its bearing on transport economy.

Tariffs are indicated for rail and road transport of irradiated fuel. Insurance cover against nuclear damage to third parties is available through insurance pools, and premiums are quoted.

An estimate of overall transport costs together with possible processing tariffs for near-natural uranium indicate that routine shipments to centrally-located processing plants would be reasonable. The influence of transport charges on the total fuel cycle cost is shown.